GEORGIA INSTITUTE OF TECHNOLOGY OFFICE OF CONTRACT ADMINISTRATION

PROJECT ADMINISTRATION DATA SHEET

[x] ORIGINAL [ ] REVISION NO. ______

Project No. E-26-686*

DATE 12-15-81

Project Director: Dr. W. M. Stacey

School/~~Lab~~ Nuclear Eng.

Sponsor: Dept. of Energy, Oak Ridge Operations

Type Agreement: Contract DE-AS05-78ET-52025, Mod A004

Award Period: From 11/1/81 To 11/30/82 (Performance) ______ (Reports)

Sponsor Amount: $125,000

Contracted through: ~~GTRI~~/GIT

Cost Sharing: N/A

Title: A Fusion Studies Program

ADMINISTRATIVE DATA

OCA Contact William F. Brown x4820

1) Sponsor Technical Contact:

C. R. Head
Systems & Aplications Studies Branch
Dept. of Energy
Washington, DC 20545

2) Sponsor Admin/Contractual Matters:

Ms. Joyce Carringer
Procurement & Contracts Division
Dept. of Energy
Oak Ridge Operations
P. O. Box E
Oak Ridge, TN 37830
(615) 576-7564

Defense Priority Rating: none

Security Classification: none

RESTRICTIONS

See Attached Gov't Supplemental Information Sheet for Additional Requirements.

Travel: Foreign travel must have prior approval – Contact OCA in each case. Domestic travel requires sponsor approval where total will exceed greater of $500 or 125% of approved proposal budget category.

Equipment: Title vests with ______

COMMENTS:

* Mod No. A004 adds $125,000 and extends contract through 11/30/82. Continuation of project E-26-670. Revised total value of contract (including prior project numbers) is $465,000.

COPIES TO:

Administrative Coordinator
Research Property Management
Accounting
Procurement/EES Supply Services

Research Security Services
~~Reports Coordinator (OCA)~~
Legal Services (OCA)
Library

EES Public Relations (2)
Computer Input
Project File
Other ______

FORM OCA 4:781

GEORGIA INSTITUTE OF TECHNOLOGY OFFICE OF CONTRACT ADMINISTRATION

SPONSORED PROJECT TERMINATION/CLOSEOUT SHEET

Date 3/28/84

Project No. E-26-686 School/~~Lab~~ NE

Includes Subproject No.(s)

Project Director(s) Dr. W. M. Stacey ~~GTRI~~ / GIT

Sponsor Dept. of Energy, Oak Ridge, TN

Title A Fusion Studies Program

Effective Completion Date: 11/30/82 (Performance) 11/30/82 (Reports)

Grant/Contract Closeout Actions Remaining:

- [X] None
- [ ] Final Invoice or Final Fiscal Report
- [ ] Closing Documents
- [ ] Final Report of Inventions
- [ ] Govt. Property Inventory & Related Certificate
- [ ] Classified Material Certificate
- [ ] Other

Continues Project No. E-26-670 Continued by Project No. E-26-611

COPIES TO:

Project Director
Research Administrative Network
Research Property Management
Accounting
Procurement/EES Supply Services
Research Security Services
Reports Coordinator (OCA)
Legal Services

Library
GTRI
Research Communications (2)
Project File
Other

Form OCA 60:1028

E-26-686

GEORGIA TECH FUSION STUDIES PROGRAM

Contract DE-AS05-78ET-52025
Quarterly Progress Report for
November 1, 1981 - January 31, 1982

Neutral Beam-Driven Impurity Flow Reversal

The principal effort over this report period has been an improvement of the computational tools in order to ultimately provide a more vigorous prediction of the potential of flow reversal for impurity control in FED/INTOR type of tokamak plasmas. The impurity peaking model has been revised (see below) to account for the ion displacement required to maintain ambipolarity, which allows for a more realistic comparison of the semi-analytical model with the PLT and ISX experiments. The new model has been confirmed by comparision with direct transport calculations. The one-dimensional transport code obtained from ORNL has been modified to couple the transport of ions and impurities. The formulation of a computation model that allows two impurity species has begun.

Sharp, central impurity peaks have been predicted by Pfirsch-Schluter transport theory, but in general no peaks have been observed in experiments. Typically, the peak was predicted by ignoring the impurity's displacement of the main ion due to ambipolarity. A simple analysis, later qualitatively confined by computer analysis, showed that this peak is much smaller. For iron, with an equilibrium peak previously projected as a factor of Z = 26 times sharper than the initial main ion profile, was found to show a peaking of approximately 3.5 when $n_z Z^2/n_i \sim 1$. The effect of small amounts of a lighter impurity such as oxygen is expected to further reduce this peak. This anlaysis may lead to a better understanding of the experimental evidence.

Ripple Reduction Poloidal Field Coils for Tokamak Reactors

A ripple reduction poloidal field (RRPF) coil design was presented at the Ninth Symposium on Engineering Problems of Fusion Research.[1] For a tokamak reactor with INTOR dimensions but only eight toroidal field coils, local magnetic ripple was reduced from more than 2% to less than 0.1% throughout the plasma. The poloidal field can be adjusted independent of the ripple reduction to produce an applied poloidal field suitable for a mildly elongated, D-shaped plasma equilibrium. Studies have indicated that lack of axisymmetry appears to be no problem for this partiuclar coil arrangement.

Work is in progress to design RRPF coils for tokamak reactors with smaller toroidal field coils and to include these RRPF coils in the same vacuum Dewar and cryogenic structural support as the toroidal field coils.

Ripple-Induced Transport

The 2-D ripple transport code has been expanded to correct for constraints which can be applied to the ripple trapped (RT) transport mechanism. The first of these constraints is the localized evaluation of dB/ds to determine the existence of local ripple wells. Failure to have a local well results in no contribution to the surface integrated transport. The second improvement is the evaluation of the effective ripple well depth, δ_{eff}. This quantity is now properly used in the loss calculations in place at the ripple well depth, δ_o. The necessary theoretical support

---

1 G. Bateman, "Ripple Reduction Poloidal Field Coils for Tokamak Reactors," Ninth Symposium on Engineering Problems of Fusion Research, Chicago, 26-29 October 1981; Georgia Tech Fusion Report GTFR-28 (October 1981).

for the change has also been done. The third improvement is the evaluation of the collisionality constraints that are placed upon the ripple trapped particles. These constraints will reduce the number of ripple trapped particles due to the minimum $V_{\parallel}$ requirement and the maximum V requirement. The transport calculations are corrected for the change in the distribution function, f^{-1}, of particles that experience the ripple trapped mechanism. Previously, the mechanism was considered to be either totally on or totally off due to evaluation of the collisionality constraints and the thermal velocity, V_{th}. Testing of these improvements has yielded good results.

At present, the code is being expanded to include other ripple-related transport mechanisms, notably Banana drift and ripple plateau. The theoretical development has already been done, and the algorithms are being developed.

E-26-686

GEORGIA TECH FUSION STUDIES PROGRAM

Contract DE-AS05-78ET-52025
Quarterly Progress Report
February 1, 1982 - April 30, 1982

HYBRID RRPF COIL DESIGN

Previous Ripple Reduction Poloidal Field (RRPF) coil designs (see reports GTFR-26 and -28) required a set of superconducting RRPF coils off the outboard edge of the plasma, between the shielding and the toroidal field (TF) coils, to reduce ripple near the plasma midplane as well as a set of RRPF coils above and below the shielding to reduce ripple near the top and bottom of the plasma and to provide the appropriate poloidal field shaping. The outboard RRPF coils near the midplane made it difficult to access sectors of the shielding, blanket, and first wall for replacement. A new conceptual design has been developed which solves this problem by using normal saddle coils imbedded in the shielding to reduce ripple near the plasma midplane together with superconducting RRPF coils above and below the shielding to reduce ripple near the top and bottom of the plasma and provide the poloidal field for a D-shaped plasma equlibrium. In an 8 TF coil design, each saddle coil needs to carry less than 500 kA turns of current and the entire set consumes less than 7 MW of power. The design for an FED/INTOR like tokamak reactor provides clearance for a minimum of 0.8 m of shielding so that organic insulation can be used. Current in the cross legs of the saddle coils are balanced to produce no net poloidal field. The RRPF coils share the TF coil cryostat and structural support, completely above and below the shielding.

Access with 6 and 8 TF Coils

RRPF coils can be used to design tokamak reactors with fewer and smaller TF coils. With 6 TF coils, sectors of shielding, blanket and first wall can be pulled out easily along lines tangent to the TF coils. However, twice as much current (.92 MA turns) and 3.4 times as much power (22 MW) is required to drive the saddle coils, and more than three times as much net current (6.8 MA turns) is required in the radial legs of the

RRPF coils, compared to the reference 8 TF coil design. Consequently, while the 6 TF coil design may be possible, the 8 TF coil design is much easier and should be implemented first.

Access is more of a problem in the 8 TF coil design. For the optimum configuration, it was found best to pull out sectors along a line 55° relative to the nearest TF coil (10° off the tangent to the next TF coil). Using this design, the TF coils can be reduced to a 6.6 m by 9.2 m bore. The RRPF and saddle coils reduce magnetic ripple from more than 6% to less than 0.6% at the outer edge of the plasma, and much less within the plasma. The water cooled copper saddle coils are relatively easy to remove while sectors are being replaced.

A detailed design using overlapping coils is being developed in order to keep the maximum field at the surface of the superconducting RRPF coils at or below 8 tesla. These coils need to be pulsed during the plasma start-up in order to accommodate the changes needed in the poloidal field shaping. Since only a slow change may be needed in the poloidal shaping field, especially if an RF driven current start-up is used, the 8 tesla restriction may be relaxed in the future.

Plasma Equilibrium with RRPF Coils

D-shaped equilibria with a separatrix at the edge of the plasma, suitable for a poloidal divertor, with beta up to several percent have been computed with the poloidal shaping field provided by RRPF coils. Since the RRPF coils are as close to the plasma as permitted by shielding requirements, they are most effective for controlling the cross sectional shape of the plasma and the position of the separatrix. One or more additional poloidal field coils, carrying less current, are used to control the net vertical field for horizontal positioning of the plasma. The combination works quite well.

The equlibrium computations were carried out with the axisymmetric average poloidal field from the RRPF coils, using a Princeton equilibrium code. Since the saddle coils are placed relatively close to the plasma, with a minimum distance of 0.8 m to the first wall, there was some concern that the non-axisymmetric part of the field would produce magnetic islands or lack of field line confinement. This problem was examined once again by using a field line following code with the complete vacuum field from all the coils together with a filamentary approximation to the plasma current. In all of the many runs following field lines several hundred times around, only simply nested magnetic surfaces were observed to within one part in 10^4. Hence, the non-axisymmetry of the RRPF and saddle coils does not seem to be a problem.

FLOW REVERSAL

Considerable progress has been made in benchmarking the model which will be used for evaluating the feasibility of impurity flow reversal for commercial tokamaks. The PLT flow reversal experiment has been analyzed. A model has been developed which can predict the measured impurity fluxes without neutral beam injection and with co-injected beams, thus predicting the observed flow reversal. Unfortunately, the model does not yet predict the magnitude of the enhanced inward impurity flux with counter injection. A conjecture that this latter difficulty may be associated with toroidal rotation effects will be explored.

E-26-686

GEORGIA TECH FUSION STUDIES PROGRAM

Contract DE-AS05-78ET-52025
Quarterly Progress Report
4 October 1982

RIPPLE REDUCTION COILS FOR TOKAMAK REACTORS

A conceptual design is presented using a combination of superconducting Ripple Reduction Poloidal Field (RRPF) coils [1,2] and normal saddle coils in an INTOR-size tokamak reactor consisting of 8 toroidal field (TF) coils with 6.6 m by 9.2 m bore. A cross section of the design is shown in Figure 1. The RRPF coils consist of dipole magnetic coils placed between the TF coils and the neutron shielding, 3.8 m above and below the plasma midplane, in order to produce part of the shaping field needed for a D-shaped plasma equilibrium and to reduce magnetic ripple near the top and bottom of the plasma. They are designed to share the TF coil cryostat and support structure, without obstructing access to the shielding, blanket or first wall. The normal (ie, water cooled copper) saddle coils are attached to the outer edge of the shielding, one meter away from the first wall. They carry 0.6 MA turn each and consume less than 16 MW power all together to reduce magnetic ripple primarily near the midplane of the plasma. A field line following code was used to show that magnetic field lines are well confined in spite of the lack of axisymmetry in the coil currents.

In this design with 8 moderately sized TF coils, magnetic ripple is reduced from more than 5% at the outer edge of the plasma to less than 0.3% throughout the plasma. The TF coil size was determined so that each sector of the shielding, blanket and first wall could be pulled straight out along a line 55 degrees relative to the nearest TF coil. With a six TF coil set of the same size, access is easier but more than twice as much current is needed in the saddle coils to reduce ripple.

Since the superconducting RRPF coils are intended to be pulsed, in response to changing plasma equilibrium conditions, they must be designed so that the maximum field strength at their surface is everywhere less than 8 tesla. This constraint poses quite a challenge since the ambient toroidal field exceeds 7 tesla at the inner edge of the RRPF coils. The challenge was met with the following design: Each RRPF coil carrying 1.6 MA turns of current is 0.1 m thick, 0.8 m wide, and extends two and one half toroidal sectors (112.5 degrees) around the torus. These thin coils overlap each other alternately two and three deep so that the average current in the toroidal legs is 4.0 MA turn. The coils, with semicircular radial legs, are staggered between the TF coils to reduce the ripple by 1.4% near the top and bottom edge of the plasma. The EEFI code was used to show that the maximum field at the RRPF coil surface at full current is about 8 tesla. The RRPF coils in this design are primarily responsible for shaping the plasma equilibrium, while the saddle coils provide independent control over the magnetic ripple.

The Princeton equilibrium code was used to determine what coil currents are needed to produce appropriate axisymmetric plasma equilibria. The RRPF coils alone produce much of the vertical field needed for positioning the plasma along the major radius as well as hexapole and some quadrupole field needed for a D-shaped plasma. With the RRPF coil current fixed, additional poloidal field coils placed outside the TF coils were used to control the plasma position and elongation. It was found that less than 2.6 MA turns of current is needed in the elongation PF coil (at z=6.2 m and R=4.6 m) to produce a plasma elongation exceeding 1.8 at the separatrix and 1.4 near the magnetic axis. The outer PF coil (at z=4.0 m and R=10.0 m) needs to carry about 3.2 MA turns of current in order to hold a plasma with 5.1% average beta. With fixed RRPF coil current, the plasma shape is remarkably insensitive to changes in the elongation PF coil current.

DEFINITION OF TECHNOLOGY DEVELOPMENT NEEDS

As an outgrowth of other activities, the preparation of a review paper on the definition of technology development needs has been initiated. One of the principal investigators (W.M. Stacey, Jr.) is collaborating with investigators at other institutions in the review of reactor conceptual design studies of tokamaks, tandem mirrors and other confinement concepts to determine how well the technology development needs can be specified

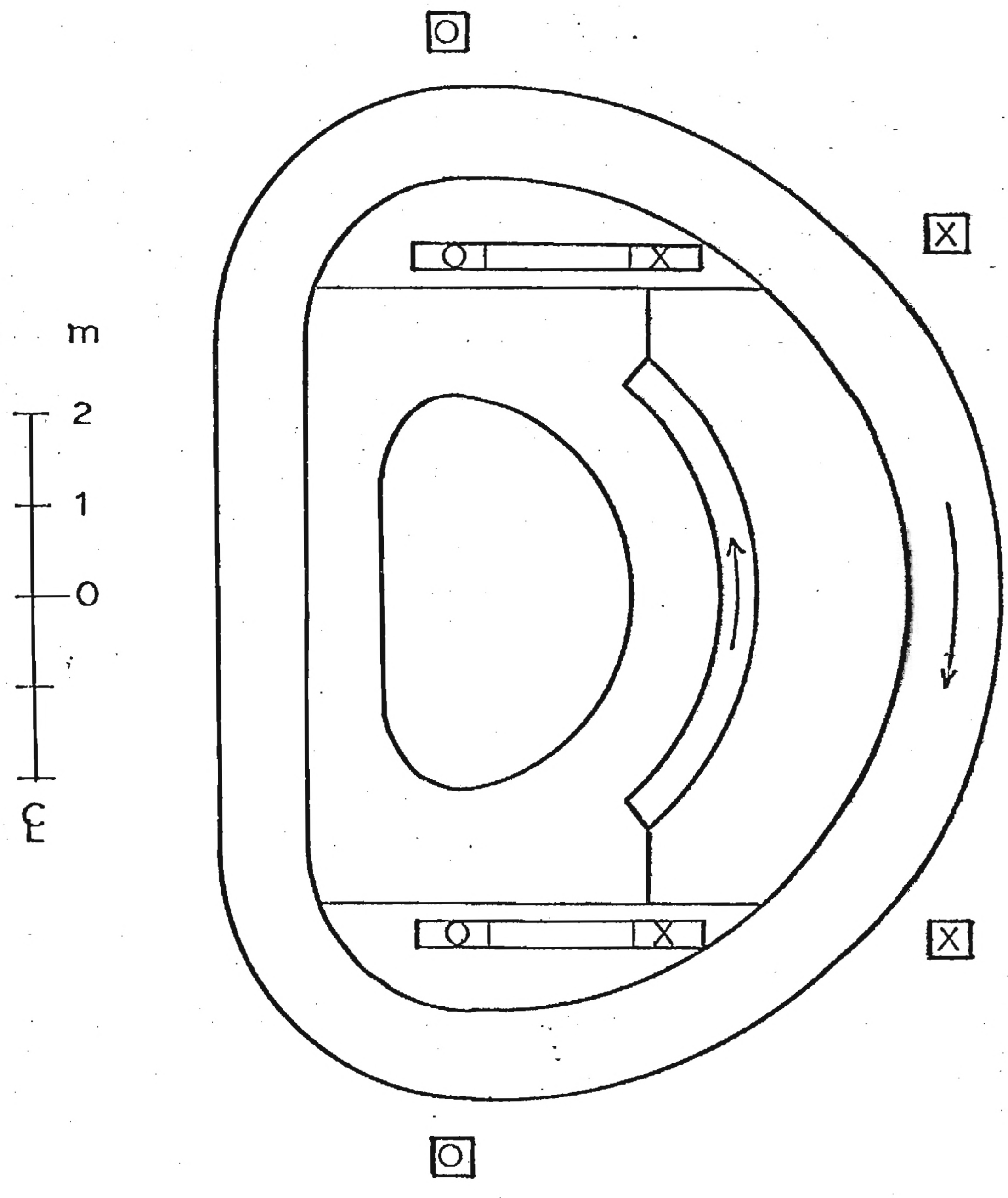

Figure 1. Cross section of tokamak reactor with D-shaped plasma chamber surrounded by blanket and shielding with saddle coils on the outboard edge, RRPF coils above and below, TF coils, and additional PF coils.

on the basis of the relative maturity of both the confinement concept and the conceptual design effort, and to evaluate the degree of commonality among the technology needs for the different confinement concepts.

References

1. G. Bateman, "Ripple Reduction Poloidal Field Coils for Tokamaks", Georgia Tech Fusion Report GTFR-26 (March, 1981).
2. G. Bateman, "Ripple Reduction Poloidal Field Coils for Tokamak Reactors", Ninth Symposium on Engineering Problems of Fusion Research, Chicago, 26-29 October 1981; and Georgia Tech Fusion Report GTFR-28 (October, 1981).

F26-686

undertake if a supplemental budget is available to support a post-doctoral position. (3) The third additional task which we propose is the initiation of a model that can simulate the interactive dynamics of the plasma and the technological and control systems in a tokamak fusion reactor.

II. RIPPLE REDUCTION POLOIDAL FIELD COILS (G. Bateman)

A. Background

Ripple Reduction Poloidal Field (RRPF) coils [1,2] consist of dipole magnetic coils placed between the toroidal field (TF) coils of a tokamak reactor in order to produce the poloidal magnetic field needed for plasma equilibrium and at the same time to reduce magnetic ripple from the toroidal field coils. Since they can be placed close to the plasma, without linking the TF coils, RRPF coils require less current to hold and shape the plasma than the presently envisioned PF coil systems [3-6] which they would augment or replace. There would consequently be less stress on the TF coils; less structural support and smaller power supplies would be needed. RRPF coils are relatively small and modular, easier to manufacture, transport, access and replace that conventional PF coils. Since they are closer to the plasma, their shaping fields can be designed to produce a more stable plasma shape and separatrix position.

Since RRPF coils can be designed to reduce magnetic ripple as needed, independent of the poloidal field they produce, tokamak reactors can be designed with fewer TF coils, for greater access, and with smaller TF coils for reduced cost, closer conventional PF coils, further reducing coil stress and structural support. Since the exact effect of magnetic ripple in a tokamak reactor is not known, RRPF coils can be used to increase the certainty that the first generation of tokamak reactors will reach ignition. They can also be used to vary the strength and spatial extent of the magnetic ripple over a wide range for fusion burn control.

B. Previous Work Under Contract

It has been demonstrated that RRPF coil configurations can be designed to reduce magnetic ripple to extremely low levels [1,2]. For example, ripple was reduced from more than 2% at the outer edge of the plasma to less than 0.1%

throughout the plasma in an 8 TF coil configuration with 8 m by 11.4 m inner bore and Phase Zero INTOR dimensions. [3] In a more recent design, ripple was reduced from more than 15% to less than 1% in a configuration with only 6 TF coils and with smaller dimensions (6.4 m by 9.2 m inner bore). Recent work has concentrated on configurations with 8 TF coils, since 8 TF coil designs have been developed with adequate access to the shielding, blanket and first wall, and since the corresponding 6 TF coil designs require more than twice the current in the RRPF coils.

It has been been demonstrated that D-shaped plasma equilibria with several percent beta and a separatrix within the first wall can be produced with RRPF coils alone or, alternatively with a combination of RRPF coils together with conventional PF coils outside the TF coil set. RRPF coils above and below the plasma, just outside the neutron shielding, are most effective at controlling the position of the separatrix and the plasma elongation, while a modest current in the exterior PF coil provides independent control of the vertical field needed to position the plasma along the major radius. A field line following code has been used to test each new configuration, but no evidence has been found so far for loss of field line confinement due to lack of axisymmetry.

Over the past six months, a new conceptual design has been developed using a combination of normal (e.g. water cooled copper) saddle coils [7-10] imbedded in the neutron shielding to reduce ripple near the plasma midplane together with superconducting RRPF coils above and below the shielding to reduce ripple near the top and bottom of the plasma and to provide the poloidal field needed for a D-shaped plasma equilibrium. In an 8 TF coil design, each saddle coil carries less than 0.5 MA turns of current and the entire set consumes less than 10 MW of power. The current in the cross legs of the saddle coils is balanced so that they produce no net poloidal field. The superconducting RRPF coils share the TF coil cryostat and structural support. Since they are located completely above and below the shielding, they do not inhibit access to the tokamak. There is enough space between the TF coil return legs to pull out one-eigth sectors of the shielding, blanket, and first wall.

C. Proposed Effort for the Next Contract Period

Significant improvements have been made in RRPF coil design over the past year for FED/INTOR like tokamak reactors [3-5]. During the next year, we will examine the potential advantages of using RRPF coils and similar systems in larger tokamak reactors such as STARFIRE [6]. A variety of options will be explored using combinations of RRPF coils, saddle coils [7-10], and ferro-magnetic shielding [11] in an effort to maximize access to the blanket and shielding sectors, maximize control over the ripple and plasma shape, and minimize cost, stresses, power requirements and complexity. Much work needs to be done on assessing the impact of these coils on the overall system design. For example, since the constraint on ripple is no longer a problem, does the advantage of reducing TF coil size, together with reduced power supply and structural requirements, outweigh the advantage of maintaining access to complete shielding and blanket sectors? Would it be better to increase the plasma size for improved confinement and lower beta, with a slight increase in total machine size?

It has been demonstrated that RRPF coils alone, or together with external PF coils, can produce a D-shaped plasma with the separatrix within the first wall. Since RRPF coils are closer to the plasma than the PF coils, it should be possible to use RRPF coils to control the position of the separatrix with better vertical stability. A three dimensional equilibrium code is needed to determine the effect of the non-axisymmetric RRPF coil fields on the shape of the separatrix and scrape-off layer. New concepts, such as a "stitch divertor" will be explored, in which the scrape-off layer is pulled out into the divertor in only a portion of each toroidal sector. The resulting divertor may have better vertical stability and may require less coil current.

Work will continue on improving the detailed coil design, to ensure the feasibility of using superconducting RRPF coils. With each new design, further tests will be made of field line confinement, plasma equilibrium, and the effects of reductions in and spatial distribution of the magnetic ripple.

References for Task 1

1. G. Bateman, "Ripple Reduction Poloidal Field Coils for Tokamaks", Georgia Tech Fusion Report GTFR-26 (March, 1981).

2. G. Bateman, "Ripple Reduction Poloidal Field Coils for Tokamak Reactors", Ninth Symposium on Engineering Problems of Fusion Research, Chicago, 26-29 October 1981; and Georgia Tech Fusion Report GTFR-28 (October, 1981).
3. INTOR, Phase Zero, IAEA, Vienna, 1980.
4. INTOR, Phase One Conceptual Design, IAEA, Vienna, 1982.
5. Fusion Engineering Device (FED), DOE/TIC-11600 (October, 1981).
6. STARFIRE, ANL/FPP-80-1 (September 1981).
7. P.F. Michaelson, S.O. Hong, W.C. Wong, I.N. Sviatoslavsky, and R.W. Conn, "A Toroidal Field Magnet System Utilizing Normal Metal Trimming Coils", UWFDM-225 and Proceedings of the Seventh Symposium on Engineering Problems of Fusion Research, Knoxville, Tenn., October, 1977.
8. B. Badger et. al., "NUWMAK, A Tokamak Reactor Design Study", UWFDM-330, (March, 1979).
9. R.J. Thome and J.M. Tarrh MHD and Fusion Magnets; Field and Force Design Concepts, Wiley, 1982.
10. N. Ohyabu, "New Toroidal Coil System for Tokamaks," General Atomic Report, GA-A14874 (June, 1979).
11. L.R. Turner, S-T. Wang and H.C. Stevens, "Iron Shielding to Decrease Toroidal Field Ripple in a Tokamak Reactor," Proceedings of the Seventh Symposium on Engineering Problems of Fusion Research, Knoxville, Tenn., October, 1977.

III. IMPURITY FLOW REVERSAL

A. Background

When toroidal momentum is injected into a tokamak by neutral beams, the radial transport of impurities can be altered by three separate effects:

1. There is a direct effect[1)] of momentum exchange between the fast beam ions and the impurity ions, which is analogous to interspecies friction in its effect upon radial transport.
2. There is an indirect effect[2)] of the external momentum input (and compensating external drags) on the plasma momentum balance and hence on the particle flows in the flux surfaces and consequently upon the radial transport; and

3. There is an inertial effect[3)] which arises because the $\vec{V}\cdot\nabla\vec{V}$ term which must be retained in the momentum balance when the toroidal plasma velocity becomes of the same order as V_{th}, giving rise to a modification of the radial transport.

The first two effects are referred to as the direct momentum coupling effect, and the last effect is referred to as the inertial effect.

Based upon rough estimates[2)] that co-injection should tend to drive impurities outward in ISX and PLT and that the effects should be large enough to be observed, flow reversal experiments were performed in those devices[4,5)]. In both cases, impurity accumulation in the center was much greater with counter-injection than with co-injection, in qualitative agreement with the theory. Similar results were obtained in T-10[6)].

The implication of these results is that the coinjection of momentum by neutral beam injection (or by rf) is a potential active impurity control mechanism for tokamaks. Such a mechanism would complement the pumped-limiter, which pumps helium but generates impurities at the plasma by sputtering from the limiter, and the combination could potentially be competitive with the poloidal divertor.

B. Previous Work Under Contract

1. The Stacey-Sigmar formalism for the direct coupling effect has been extended to include temperature gradient effects (GTFR-21), reduced to a computational form and incorporated into a 1D transport code.
2. A preliminary assessment (GTFR-25) was made of the NBI power requirements for impurity flow reversal in FED/INTOR plasmas. Based on rather crude models, it was estimated that ~25-50 MW would be sufficient to prevent edge-generated impurities from entering the plasma.
3. A careful analysis of the PLT flow reversal experiment[4)] is being performed during the present contract period. To date, it has been possible to develop a model based upon the Pfirsch-Schluter, gradient-driven flux and the Stacey-Sigmar NBI-driven flux which underpredicts both the inward impurity fluxes without beam injection and the outward impurity fluxes with coinjection by a factor of ~2. This model underpredicts the inward fluxes with counterinjection by ~4. The present formulation[3] of the inertial effects is clearly imcompatible with

the experiment, although the magnitude is consistent, which implies a need to improve the consistency of the formulation.

4. The model which predicts the PLT flow-reversal with coinjection, 2x (P-S+S-S), has been used to estimate the NBI power requirements for flow reversal in FED/INTOR type plasmas. It is estimated that 10-30 MW may be sufficient. This work is being extended to allow the impurity distribution to evolve from a fixed initial profile, which has been used to date, to an equilibrium distribution.
5. Calculations have been initiated to estimate the NBI power required for flow reversal in commercial tokamak reactors. Estimates will also be made of possible flow reversal effects in TFTR.

C. Proposed Effort for the Next Contract Period

1. The assessment of NBI-driven impurity flow reversal for ETR/INTOR and commercial tokamak reactors will be completed. This will include an attempt to incorporate a consistent formulation of the inertial effect in the model and transport modeling of time-dependent impurity fluxes. A systematic survey of possible commercial tokamak plasma conditions and the possibility of the development of high-energy negative ion beams will be included in the evaluation of the ultimate feasibility of NBI-driven impurity flow reversal for commercial tokamak reactors.
2. The momentum-driven impurity flow reversal formalism will be extended to treat momentum input from ICRF and LHR waves. The possibility of confirming this theory in present experiments will be examined. An assessment of rf-driven impurity flow reversal for ETR/INTOR and commercial tokamak reactors will be initiated.
3. There is experimental evidence[7)] that viscous, radial momentum transfer in a rotating plasma is much larger than had heretofore been thought. Such a process, which has been ignored previously, would also affect impurity flow reversal by NBI. The theoretical formalism for treating this effect is being developed under other support. The effort proposed under this contract is to reduce the formalism to computational form, incorporate this effect into the existing flow reversal impurity transport code and to upgrade the NBI flow reversal assessment for ETR/INTOR and commercial type plasmas.

4. There are indications (J.A. Schmidt, private communication) that a tokamak plasma with a pump-limiter will have a large poloidal gas flow in the scrape-off region. This will effect the poloidal flows in the outer plasma region and consequently alter the impurity transport in the outer regions of the plasma. It is proposed to extend the Stacey-Sigmar formalism to treat this effect and to assess the potential for this type of impurity flow reversal in ETR/INTOR and commercial type plasmas.

References for Section III

1. T. Ohkawa, Kakuyogo Kenkyu, 32, 1 (1974).
2. W.M. Stacey, Jr. and D.J. Sigmar. Phys. Fluids. 22, 2000 (1979) and Nucl. Fus., 19, 1665 (1979).
3. K.H. Burrell, T. Ohkawa and S.K. Wong, General Atomic Report GA-A16082 (1980).
4. D.R. Eames, Ph.D. Thesis, Princeton Univ. (1980).
5. R.C. Isler, et al., Plasma Physics and Controlled Nuclear Fusion Research 1980, IAEA, Vienna (1981).
6. A.B. Berlizov et al., ibid.
7. S. Suckewer, et al., Nucl. Fusion, 21, 981 (1981).

IV. PLASMA -WALL KINETICS MODEL

A. Background

The plasma-wall interaction is quite complex. The surface concentrations of the individual components of the alloy comprising a wall or limiter vary with time under the action of sputtering by energetic particles from the plasma. Preferential component sputtering, displacement mixing, bulk diffusion and segregation all act to alter with time the alloy composition on the surface exposed to the plasma. Furthermore, there will exist an influx to the wall or limiter surface of material which has been eroded from that or another surface, which in turn affects the wall kinetics and the sputtering mechanics. This redeposition influx depends upon the local plasma transport in the wall and limiter region.

Thus, an understanding of plasma-wall kinetics requires coupling the local plasma transport (redeposition), sputtering mechanics and the subsurface wall kinetics. Development of such a model is currently in progress, under other support, as the Ph.D. research of A.B. DeWald. This model includes: (1) a multi-layer, multi-component version of the TRIM Monte-Carlo code for the sputtering mechanics; (2) a subsurface wall kinetics model which treats displacement mixing, thermal and radiation diffusion, and Gibbsion and radiation-induced segregation; and (3) a wall region plasma transport model which treats cross-field and along field transport, and a sheath potential with non-normal field lines intersecting the surface, taking into account ion reflection and secondary ion emission. This model, and the Ph.D. dissertation of DeWald, should essentially be complete by mid-1983.

B. Prepared Effort for the Next Contract Period

This coupled plasma-wall kinetics model represents a unique capability for studying many aspects of the plasma-wall interaction which have a critical impact on the engineering design of first-wall and impurity control systems for tokamaks (and for other concepts, as well). We propose that DOE support a post-doctoral position for DeWald to employ this model, and extensions thereof, to perform in-depth and systematic analyses pertaining to several important topics in the plasma-wall interaction area. Among these are:

1. Erosion lifetime of limiters and divertor collector plates. The net erosion of a limiter or divertor collector plate is the difference between large erosion and redeposition rates.
2. Coatings. Thin, low-Z coatings on metal substrates have been proposed for a number of first-wall and limiter applications. The diffusion to the surface and subsequent sputtering of the substrate material and the in-situ recoating are important questions relating to the feasibility of the coating concept.
3. Characterization of wall surface composition evolution in time and consequent impurity influx to plasma for possible wall materials are needed.

Since the proposed work would not begin until mid-1983, we suggest that priorities be determined, in consultation with DOE, at that time.

V. TOKAMAK FUSION BURN CONTROL

A. Background

Since the power balance in an ignited D-T fusion reactor is thermally unstable at the desired operating plasma temperature, it will be necessary to exercise some form of burn control. Enhanced transport produced by magnetic ripple[1-4] or pressure driven instabilities such as ballooning modes[5] provide the most promising mechanisms for fusion burn control. Other transport mechanisms generally do not increase fast enough with temperature to be useful.

Different transport mechanisms will produce loss of plasma energy through different channels. For example, ripple is expected to have the largest effect on high energy particles such as new-born alpha particles or neutral beam injected ions. Energy is lost predominantly through the electron channel during the observed degradation of confinement at the highest beta values achieved in tokamak experiments. Both ripple transport and pressure driven ballooning instabilities are expected to deposit energy and particles in a nonuniform way on the wall or into the divertor.

B. Work to Date Under Contract

Our recent effort in this area started with a scoping study which determined that variable field ripple control is feasible for tokamak reactors[1]. The theory of heat transport due to ripple trapped particles was then extended to consider arbitrary magnetic flux surface geometry[2]. In particular, poloidally asymmetric and non-separable ripple profiles were allowed. A computer code was then written to calculate ripple transport coefficients from this ripple-trapped theory.

Next, the banana - drift and ripple-plateau heat transport theories were extended to consider arbitrary magnetic surface geometry and arbitrary ripple contours[3] and the collisionablity constraints were properly taken into account[4]. These improved theories and computer codes are now being used to determine the impact of magnetic ripple on tokamak fusion reactors.

C. Proposed Effort for Next Contract Year

The effect of various configurations of magnetic ripple on burn control needs to be investigated. First, vertically asymmetric ripple will be considered. This is directly applicable to the FED/INTOR design with a single null poloidal divertor in which the plasma is offset from the midplane and, therefore, particles experience a vertically asymmetric ripple from the toroidal field coils. Next, the effect of ripple controlled independently near the midplane, near the top or near the bottom of the plasma by Ripple Reduction Poloidal Field Coils[6,7] will be assessed. The effect of toroidal periodicity or a toroidally localized magnetic ripple will then be considered.

Finally, the engineering designs needed to handle the spatial distribution of power deposited on the wall or in the divertor (i.e. toward the outer or inner edge of the plasma) will be considered for both ripple enhanced and ballooning-mode-enhanced plasma losses.

References

1. J.N. Davidson and W.M. Stacey, Jr., "The Use of Field Ripple for Burn Phase Control in Short Pulse Tokamak Reactors," GTFR-9, June 1979.
2. J.N. Davidson and A. Engel, "Ion Heat Transport for Ripple Trapped Particles in Arbitrary Ripple Field Geometry," GTFR-22, February 1981.
3. J. Narl Davidson and Anthony Engel, "Modification of Banana Drift Transport Coefficients for General Ripple Contour Mappings," *GTFR-32, May 1982.
4. Anthony Engel, "Examination of Collisionality Constraints Upon Ripple Related Transport Mechanism," *GTFR-33, May 1982.
5. B.A. Carreras, J.A. Holmes, W.A. Cooper, P.H. Diamond and P.L. Similon, "Resistive Pressure Driven Modes in High Beta Tokamaks," Controlled Fusion Theory Conference, Sante Fe, April 1982.
6. Glenn Bateman, "Ripple Reduction Poloidal Field Coils for Tokamaks," *GTFR-26, March 1981.
7. Glenn Bateman, "Ripple Reduction Poloidal Field Coils for Tokamak Fusion Reactors," *GTFR-28, October 1981.

IV. TOKAMAK FUSION REACTOR DYNAMICS

A. Background

Simulation of overall system dynamics has been a valuable design tool for fission reactors since the earliest days of their development. Modeling of the interactive dynamics of the physics and the various technological and control systems was based upon lumped-parameter models that were adjusted to experiment or more detailed calculations. Such simulations were quite important in identifying limits on the design parameters of individual components or systems that would lead to a safe and acceptable dynamic performance of the overall system.

B. Proposed Work for the Next Contract Period

The initiation of the development of a lumped-parameter model for simulating the interactive dynamics of the plasma and the technological and control systems in a tokamak fusion reactor is proposed. Initial emphasis would be upon the plasma, impurity control system, first-wall and plasma-wall interaction, and the burn control system, because of our other work in these areas and because of their importance. Lumped-parameter models for these systems and their interactions would be developed and the parameters would be determined from more detailed model calculations or experiments. A computer code would be written. Studies would be initiated to determine limit on the systems parameters for satisfactory dynamic performance of a tokamak fusion reactor.

E-26-686

GEORGIA TECH FUSION STUDIES PROGRAM

Contract DE-AS05-78ET-52025
Quarterly Progress Report for
November 1, 1981 - January 31, 1982

Neutral Beam-Driven Impurity Flow Reversal

The principal effort over this report period has been an improvement of the computational tools in order to ultimately provide a more vigorous prediction of the potential of flow reversal for impurity control in FED/INTOR type of tokamak plasmas. The impurity peaking model has been revised (see below) to account for the ion displacement required to maintain ambipolarity, which allows for a more realistic comparison of the semi-analytical model with the PLT and ISX experiments. The new model has been confirmed by comparision with direct transport calculations. The one-dimensional transport code obtained from ORNL has been modified to couple the transport of ions and impurities. The formulation of a computation model that allows two impurity species has begun.

Sharp, central impurity peaks have been predicted by Pfirsch-Schluter transport theory, but in general no peaks have been observed in experiments. Typically, the peak was predicted by ignoring the impurity's displacement of the main ion due to ambipolarity. A simple analysis, later qualitatively confined by computer analysis, showed that this peak is much smaller. For iron, with an equilibrium peak previously projected as a factor of Z = 26 times sharper than the initial main ion profile, was found to show a peaking of approximately 3.5 when $n_z Z^2/n_i \sim 1$. The effect of small amounts of a lighter impurity such as oxygen is expected to further reduce this peak. This anlaysis may lead to a better understanding of the experimental evidence.

Ripple Reduction Poloidal Field Coils for Tokamak Reactors

A ripple reduction poloidal field (RRPF) coil design was presented at the Ninth Symposium on Engineering Problems of Fusion Research.[1] For a tokamak reactor with INTOR dimensions but only eight toroidal field coils, local magnetic ripple was reduced from more than 2% to less than 0.1% throughout the plasma. The poloidal field can be adjusted independent of the ripple reduction to produce an applied poloidal field suitable for a mildly elongated, D-shaped plasma equilibrium. Studies have indicated that lack of axisymmetry appears to be no problem for this partiuclar coil arrangement.

Work is in progress to design RRPF coils for tokamak reactors with smaller toroidal field coils and to include these RRPF coils in the same vacuum Dewar and cryogenic structural support as the toroidal field coils.

Ripple-Induced Transport

The 2-D ripple transport code has been expanded to correct for constraints which can be applied to the ripple trapped (RT) transport mechanism. The first of these constraints is the localized evaluation of dB/ds to determine the existence of local ripple wells. Failure to have a local well results in no contribution to the surface integrated transport. The second improvement is the evaluation of the effective ripple well depth, δ_{eff}. This quantity is now properly used in the loss calculations in place at the ripple well depth, δ_o. The necessary theoretical support

---

[1] G. Bateman, "Ripple Reduction Poloidal Field Coils for Tokamak Reactors," Ninth Symposium on Engineering Problems of Fusion Research, Chicago, 26-29 October 1981; Georgia Tech Fusion Report GTFR-28 (October 1981).

for the change has also been done. The third improvement is the evaluation of the collisionality constraints that are placed upon the ripple trapped particles. These constraints will reduce the number of ripple trapped particles due to the minimum $V_{\parallel}$ requirement and the maximum V requirement. The transport calculations are corrected for the change in the distribution function, f^{-1}, of particles that experience the ripple trapped mechanism. Previously, the mechanism was considered to be either totally on or totally off due to evaluation of the collisionality constraints and the thermal velocity, V_{th}. Testing of these improvements has yielded good results.

At present, the code is being expanded to include other ripple-related transport mechanisms, notably Banana drift and ripple plateau. The theoretical development has already been done, and the algorithms are being developed.

E-26-686

GEORGIA TECH FUSION STUDIES PROGRAM

Contract DE-AS05-78ET-52025
Quarterly Progress Report
February 1, 1982 - April 30, 1982

HYBRID RRPF COIL DESIGN

Previous Ripple Reduction Poloidal Field (RRPF) coil designs (see reports GTFR-26 and -28) required a set of superconducting RRPF coils off the outboard edge of the plasma, between the shielding and the toroidal field (TF) coils, to reduce ripple near the plasma midplane as well as a set of RRPF coils above and below the shielding to reduce ripple near the top and bottom of the plasma and to provide the appropriate poloidal field shaping. The outboard RRPF coils near the midplane made it difficult to access sectors of the shielding, blanket, and first wall for replacement. A new conceptual design has been developed which solves this problem by using normal saddle coils imbedded in the shielding to reduce ripple near the plasma midplane together with superconducting RRPF coils above and below the shielding to reduce ripple near the top and bottom of the plasma and provide the poloidal field for a D-shaped plasma equlibrium. In an 8 TF coil design, each saddle coil needs to carry less than 500 kA turns of current and the entire set consumes less than 7 MW of power. The design for an FED/INTOR like tokamak reactor provides clearance for a minimum of 0.8 m of shielding so that organic insulation can be used. Current in the cross legs of the saddle coils are balanced to produce no net poloidal field. The RRPF coils share the TF coil cryostat and structural support, completely above and below the shielding.

Access with 6 and 8 TF Coils

RRPF coils can be used to design tokamak reactors with fewer and smaller TF coils. With 6 TF coils, sectors of shielding, blanket and first wall can be pulled out easily along lines tangent to the TF coils. However, twice as much current (.92 MA turns) and 3.4 times as much power (22 MW) is required to drive the saddle coils, and more than three times as much net current (6.8 MA turns) is required in the radial legs of the

RRPF coils, compared to the reference 8 TF coil design. Consequently, while the 6 TF coil design may be possible, the 8 TF coil design is much easier and should be implemented first.

Access is more of a problem in the 8 TF coil design. For the optimum configuration, it was found best to pull out sectors along a line 55° relative to the nearest TF coil (10° off the tangent to the next TF coil). Using this design, the TF coils can be reduced to a 6.6 m by 9.2 m bore. The RRPF and saddle coils reduce magnetic ripple from more than 6% to less than 0.6% at the outer edge of the plasma, and much less within the plasma. The water cooled copper saddle coils are relatively easy to remove while sectors are being replaced.

A detailed design using overlapping coils is being developed in order to keep the maximum field at the surface of the superconducting RRPF coils at or below 8 tesla. These coils need to be pulsed during the plasma start-up in order to accommodate the changes needed in the poloidal field shaping. Since only a slow change may be needed in the poloidal shaping field, especially if an RF driven current start-up is used, the 8 tesla restriction may be relaxed in the future.

Plasma Equilibrium with RRPF Coils

D-shaped equilibria with a separatrix at the edge of the plasma, suitable for a poloidal divertor, with beta up to several percent have been computed with the poloidal shaping field provided by RRPF coils. Since the RRPF coils are as close to the plasma as permitted by shielding requirements, they are most effective for controlling the cross sectional shape of the plasma and the position of the separatrix. One or more additional poloidal field coils, carrying less current, are used to control the net vertical field for horizontal positioning of the plasma. The combination works quite well.

The equlibrium computations were carried out with the axisymmetric average poloidal field from the RRPF coils, using a Princeton equilibrium code. Since the saddle coils are placed relatively close to the plasma, with a minimum distance of 0.8 m to the first wall, there was some concern that the non-axisymmetric part of the field would produce magnetic islands or lack of field line confinement. This problem was examined once again by using a field line following code with the complete vacuum field from all the coils together with a filamentary approximation to the plasma current. In all of the many runs following field lines several hundred times around, only simply nested magnetic surfaces were observed to within one part in 10^4. Hence, the non-axisymmetry of the RRPF and saddle coils does not seem to be a problem.

FLOW REVERSAL

Considerable progress has been made in benchmarking the model which will be used for evaluating the feasibility of impurity flow reversal for commercial tokamaks. The PLT flow reversal experiment has been analyzed. A model has been developed which can predict the measured impurity fluxes without neutral beam injection and with co-injected beams, thus predicting the observed flow reversal. Unfortunately, the model does not yet predict the magnitude of the enhanced inward impurity flux with counter injection. A conjecture that this latter difficulty may be associated with toroidal rotation effects will be explored.

E-26-686

GEORGIA TECH FUSION STUDIES PROGRAM

Contract DE-AS05-78ET-52025
Quarterly Progress Report
4 October 1982

RIPPLE REDUCTION COILS FOR TOKAMAK REACTORS

A conceptual design is presented using a combination of superconducting Ripple Reduction Poloidal Field (RRPF) coils [1,2] and normal saddle coils in an INTOR-size tokamak reactor consisting of 8 toroidal field (TF) coils with 6.6 m by 9.2 m bore. A cross section of the design is shown in Figure 1. The RRPF coils consist of dipole magnetic coils placed between the TF coils and the neutron shielding, 3.8 m above and below the plasma midplane, in order to produce part of the shaping field needed for a D-shaped plasma equilibrium and to reduce magnetic ripple near the top and bottom of the plasma. They are designed to share the TF coil cryostat and support structure, without obstructing access to the shielding, blanket or first wall. The normal (ie, water cooled copper) saddle coils are attached to the outer edge of the shielding, one meter away from the first wall. They carry 0.6 MA turn each and consume less than 16 MW power all together to reduce magnetic ripple primarily near the midplane of the plasma. A field line following code was used to show that magnetic field lines are well confined in spite of the lack of axisymmetry in the coil currents.

In this design with 8 moderately sized TF coils, magnetic ripple is reduced from more than 5% at the outer edge of the plasma to less than 0.3% throughout the plasma. The TF coil size was determined so that each sector of the shielding, blanket and first wall could be pulled straight out along a line 55 degrees relative to the nearest TF coil. With a six TF coil set of the same size, access is easier but more than twice as much current is needed in the saddle coils to reduce ripple.

Since the superconducting RRPF coils are intended to be pulsed, in response to changing plasma equilibrium conditions, they must be designed so that the maximum field strength at their surface is everywhere less than 8 tesla. This constraint poses quite a challenge since the ambient toroidal field exceeds 7 tesla at the inner edge of the RRPF coils. The challenge was met with the following design: Each RRPF coil carrying 1.6 MA turns of current is 0.1 m thick, 0.8 m wide, and extends two and one half toroidal sectors (112.5 degrees) around the torus. These thin coils overlap each other alternately two and three deep so that the average current in the toroidal legs is 4.0 MA turn. The coils, with semicircular radial legs, are staggered between the TF coils to reduce the ripple by 1.4% near the top and bottom edge of the plasma. The EEFI code was used to show that the maximum field at the RRPF coil surface at full current is about 8 tesla. The RRPF coils in this design are primarily responsible for shaping the plasma equilibrium, while the saddle coils provide independent control over the magnetic ripple.

The Princeton equilibrium code was used to determine what coil currents are needed to produce appropriate axisymmetric plasma equilibria. The RRPF coils alone produce much of the vertical field needed for positioning the plasma along the major radius as well as hexapole and some quadrupole field needed for a D-shaped plasma. With the RRPF coil current fixed, additional poloidal field coils placed outside the TF coils were used to control the plasma position and elongation. It was found that less than 2.6 MA turns of current is needed in the elongation PF coil (at z=6.2 m and R=4.6 m) to produce a plasma elongation exceeding 1.8 at the separatrix and 1.4 near the magnetic axis. The outer PF coil (at z=4.0 m and R=10.0 m) needs to carry about 3.2 MA turns of current in order to hold a plasma with 5.1% average beta. With fixed RRPF coil current, the plasma shape is remarkably insensitive to changes in the elongation PF coil current.

DEFINITION OF TECHNOLOGY DEVELOPMENT NEEDS

As an outgrowth of other activities, the preparation of a review paper on the definition of technology development needs has been initiated. One of the principal investigators (W.M. Stacey, Jr.) is collaborating with investigators at other institutions in the review of reactor conceptual design studies of tokamaks, tandem mirrors and other confinement concepts to determine how well the technology development needs can be specified

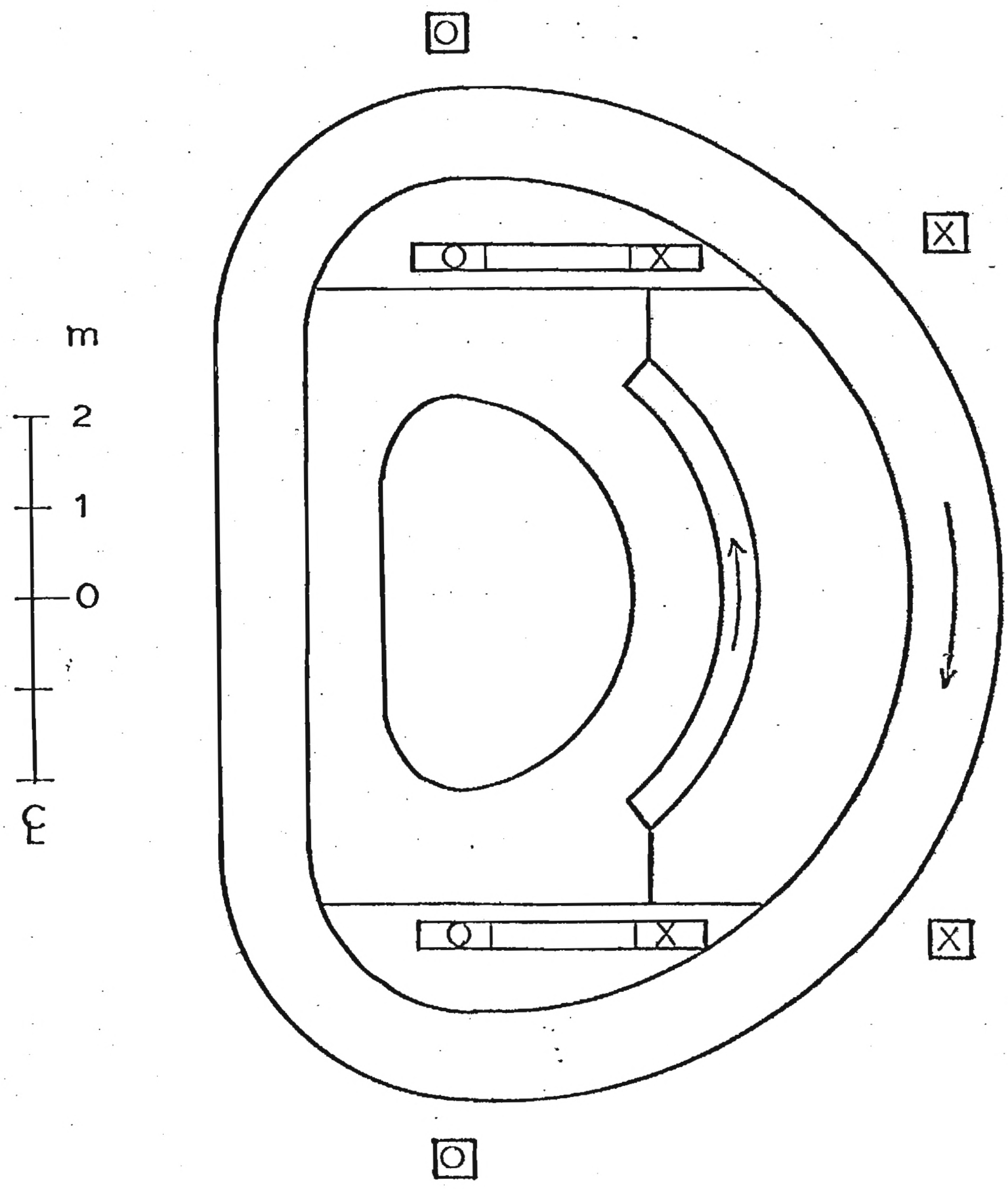

Figure 1. Cross section of tokamak reactor with D-shaped plasma chamber surrounded by blanket and shielding with saddle coils on the outboard edge, RRPF coils above and below, TF coils, and additional PF coils.

on the basis of the relative maturity of both the confinement concept and the conceptual design effort, and to evaluate the degree of commonality among the technology needs for the different confinement concepts.

References

1. G. Bateman, "Ripple Reduction Poloidal Field Coils for Tokamaks", Georgia Tech Fusion Report GTFR-26 (March, 1981).
2. G. Bateman, "Ripple Reduction Poloidal Field Coils for Tokamak Reactors", Ninth Symposium on Engineering Problems of Fusion Research, Chicago, 26-29 October 1981; and Georgia Tech Fusion Report GTFR-28 (October, 1981).